Entergy

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January 14, 2003
NL-03-009

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: Indian Point Nuclear Power Plant Unit 3
Docket No. 50-286
License No. DPR-64
Licensee Event Report # 2002-003-00

Dear Sir:

The attached Licensee Event Report (LER) 2002-003-00 is hereby submitted in accordance with the requirements of 10CFR50.73. This event is of the type defined in 10CFR50.73(a)(2)(iv) for events recorded in Entergy's corrective action process as Condition Report CR-IP3-2002-04550.

Entergy is making no new commitments in this LER. Should you have any questions regarding this submittal, please contact Mr. John McCann, Manager, Licensing, Indian Point Energy Center at (914) 734-5074.

Very truly yours,

Fred Dacimo
Vice President
Indian Point Entergy Center

cc: See next page

IE22

cc: Mr. Hubert J. Miller
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NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Indian Point Unit 3	05000- 286	1 OF 4

4. TITLE

Automatic Reactor Trip Due to the Failure of a 345 KV Main Output Breaker

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000-
11	15	2002	2002	– 003 –	00	01	14	2003	FACILITY NAME	DOCKET NUMBER 05000-

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20 2201(b)		20 2203(a)(3)(ii)		50 73(a)(2)(ii)(B)	50 73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50 73(a)(2)(iii)	50 73(a)(2)(x)
	20 2203(a)(1)		50 36(c)(1)(i)(A)	X	50 73(a)(2)(iv)(A)	73 71(a)(4)
	20 2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50 73(a)(2)(v)(A)	73 71(a)(5)
	20 2203(a)(2)(ii)		50 36(c)(2)		50 73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
	20 2203(a)(2)(iii)		50 46(a)(3)(ii)		50.73(a)(2)(v)(C)	
	20 2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50 73(a)(2)(v)(D)	
	20.2203(a)(2)(v)		50 73(a)(2)(i)(B)		50 73(a)(2)(vii)	
	20 2203(a)(2)(vi)		50 73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)	
	20 2203(a)(3)(i)		50 73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Tom Orlando, Manager, Programs & Component Engineering	(914) 735-8340

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	FK	BKR	I202	Yes					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i e., approximately 15 single-spaced typewritten lines)

On November 15, 2002, during full power operation, an automatic reactor trip (RT) occurred. The RT occurred due to the actuation of the reactor protection system (RPS), which was caused by a main turbine trip that was the result of actuation of the main generator primary and backup lockout relays. The main generator trip actuation was caused by the actuation of the electrical protective relaying direct trip circuitry from the 345 KV main output breakers. The 345 KV main output breaker No. 3 faulted open resulting in a trip signal to breakers Nos. 1, 3, and 6 which opened. Breaker 5 had opened prior to the event for unknown reasons. The protection associated with the breakers actuated the lockout relays. Station offsite power was maintained throughout the event. There was no automatic start of the Emergency Diesel Generators. The Auxiliary Feed Water pumps automatically started. The cause of the event was a RT actuation due to a failure of 345 KV main output breaker No. 3. The breaker No. 3 failure was due to high contact resistance caused by misalignment from poor vendor workmanship. Corrective actions include assessment of the output breakers, maintenance and testing of output breaker No. 1, verification that breaker No. 1 is capable of supporting 100% power, and implementation of a monitoring plan for breaker No. 1. Output breaker 3 was inspected, repaired and returned to service. The procedure for breaker maintenance will be revised. The event had no effect on public health and safety.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

DESCRIPTION OF EVENT

Note: The Energy Industry Identification System Codes are identified within the brackets {}

On November 15, 2002, at approximately 0957 hours, while at 100% steady state reactor power, an automatic reactor trip (RT) occurred. The RT occurred due to the actuation of the reactor protection system (RPS) {JC} whose logic was satisfied by a main turbine trip (TT) {JD} signal. A main turbine trip occurred as a result of the actuation of the main generator primary (86P) and backup (86BU) lockout relays {86}. The main generator trip actuation was caused by the actuation of the electrical protective relaying direct trip circuitry from the 345 KV {FK} main output breakers {BKR}. The 345 KV {FK} main output breaker {BKR} No. 3 faulted resulting in main output breakers Nos. 1, 3, and 6 opening and electrically isolating plant output. The protection associated with the breakers actuated the lockout relays (86P and 86BU). Breaker No. 5 had opened approximately one minute prior to the event for unknown reasons.

Central Control Room (CCR) {NA} operators observed the rod bottom lights, RT First Out Annunciator (Turbine Trip), and TT First Out Annunciator (Generator Primary and Backup lockout Relays) {IB}. CCR Operators then entered Emergency Operating Procedure E-0, "Reactor Trip or Safety Injection," at approximately 1000 hours, and then transitioned to procedure ES-0.1, "Reactor trip Response." Primary systems which failed to function properly was No. 32 neutron source range detector {IG} which failed low. Secondary systems that failed to function properly were the 36 circulating water pump {KE} which tripped, the 34 circulating water pump which transferred to standby drive and valve MS-PCV-1175-1 {SB} which failed to shut properly. Station offsite power was maintained throughout the event and there was no automatic start of the Emergency Diesel Generators {EK}. All rods {AA} fully inserted. The Auxiliary Feed Water System (AFWS) {BA} automatically started as expected due to steam generator level changes. Main turbine generator (MTG) overspeed occurred as expected. MTG overspeed was verified to be within limits and no unexpected or abnormal vibrations were identified. At approximately 1110 hours, CCR Operators transitioned to Plant Operating Procedure (POP) 3.1, "Plant Shutdown from 45% Power." The plant was stabilized in the hot shutdown condition and the transient terminated. At approximately 1149 hours, a four hour non-emergency notification (Incident Log No. 39375) was made to the NRC for a RPS actuation in accordance with 10CFR50.72(b)(2)(iv)(B). Operations recorded the event in the corrective action program as condition report CR-IP3-2002-04550. A post transient evaluation was performed (No. 02-02) on November 15, 2002.

Main output breaker No. 3 is a 345 KV, Type 345GA25-30, manufactured by ITE Imperial Corporation. There are six sets of contacts (fixed and moving) mounted inside each tank. The contacts are enclosed in high gas pressure chamber to quench the arc during contact disengagement between stationary and moving contacts.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

CAUSE OF EVENT

The direct cause of the event was a RT due to the actuation of the reactor protection system by a main turbine trip that was the result of actuation of the main generator primary and backup lockout relays. The main generator trip actuation was caused by the actuation of the electrical protective relaying direct trip circuitry from the 345 KV main output breakers due to the failure of main output breaker No. 3. The apparent cause of breaker No. 3 failure was the phase to ground fault due to overheating caused by high resistance at the breaker contact surfaces. The high resistance at the breaker contacts were a result of breaker contact misalignment during previous maintenance in the spring 2001 refueling outage. Misalignment of the stationary and moving contacts caused overheating. The overheating led to burning of the contacts, arcing inside the high pressure chamber causing Sulfur Hexafloride (SF6) insulating gas to lose its insulating value resulting in more arcing. Ultimately, a large arc caused a phase to ground fault resulting in a catastrophic failure of the breaker components. EPRI confirmed that operation of the breakers would not contribute any thing for misalignment. Therefore, engineering concluded the misalignment was most likely due to poor workmanship of the contract vendor. A review of vendor work determined that the same contractor that performed work on breaker No. 3 also worked on main output breaker No. 1 and 138 KV breaker No. BT5-6. Breaker No. 1 was tested and found to have elevated resistance readings on two of three phases. The cause of breaker No. 5 opening is being investigated by Con Edison.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Corrective Action Plan (CAP) to address the causes of this event and prevent recurrence.

1. Corrective maintenance was performed on main output breaker No. 1 to bring it into manufacturer's specifications, the breaker tested and returned to service.

2. The capability of main output breaker No. 1 to support 100% power was verified prior to MTG sync to the grid. The unit was synchronized to the grid on November 21, 2002.

3. A monitoring plan was developed and implemented for main output breaker No. 1.

4. Main output breaker No. 3 was inspected, cleaned and refurbished and returned to service on December 13, 2002.

5. Breaker maintenance procedure BKR-008-ELC will be revised to include verification of contact alignment and recording of contact resistances before and after contact alignment.

6. An action plan will be developed to evaluate the reliability of the SF6 gas breakers installed at Indian Point 3 (Breakers 1, 3, BT5-6).

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

EVENT ANALYSIS

The event is reportable under 10 CFR 50.73 (a) (2) (iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10 CFR 50.73 (a)(2)(iv)(B). Systems to which the requirements of 10 CFR 50.73 (a)(2)(iv)(A) apply includes the Reactor Protection System including reactor scram or reactor trip, and PWR Auxiliary or Emergency Feed Water System.

This event meets the reporting criteria because the RPS was actuated and a RT occurred due to the TT. In response to the RT the AFWS actuated as designed. After troubleshooting, corrective maintenance was performed on breaker No. 1 and the unit synchronized to the grid on November 21, 2002.

PAST SIMILAR EVENTS

A review of the past three years of Licensee Event Reports (LERs) for events that involved a RT caused by a Buchanan Substation related fault identified one event reported in LER-2000-008. LER-2000-008 reported a RT, due to a TT as a result of a generator trip initiated by the direct trip circuitry from the Buchanan Substation. The apparent cause of the direct trip was degraded electrical cable insulation between several conductors in the direct trip underground cable. The corrective actions for that event did not prevent this event because the cause was different. This event was due to a faulted main output breaker and not related to faults in the direct trip cable.

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. There was no actual safety consequences for the event because the event was an uncomplicated reactor trip with no other transients or accidents. Required safety systems performed as designed when the RT occurred. There was no risk related components out of service at the time of the RT. The AFWS actuation was expected due to steam generator level changes, which occur after automatic RT from high power levels.

There was no significant potential safety consequences of this event under reasonable and credible alternative conditions. A loss of external electrical load/turbine trip is an analyzed event described in FSAR Chapter 14. The plant performed as expected and the event was bounded by the FSAR analysis. Rod control was in automatic and a RT was initiated immediately upon a TT. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.